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Review

Overview of lead-cooled fast reactor activities

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ABSTRACT

The lead-cooled fast reactor (LFR) features a fast neutron spectrum, high temperature operation, and cooling by molten lead or lead–bismuth eutectic (LBE), low-pressure, chemically inert liquids with very good thermodynamic properties. It would have multiple applications including production of electricity, hydrogen and process heat. System concepts represented in plans of the Generation IV International Forum (GIF) System Research Plan (SRP) are based on Europe's ELFR lead-cooled system, Russia's BREST-OD-300 and the SSTAR system concept designed in the US.

The LFR has excellent materials management capabilities since it operates in the fast neutron spectrum and uses a closed fuel cycle enhanced by the fertile conversion of uranium. It can also be used as a burner to consume actinides from spent LWR fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of molten lead as a chemically inert and low-pressure coolant. In terms of sustainability, lead is abundant and hence available, even in case of deployment of a large number of reactors. More importantly, as with other fast systems, fuel sustainability is greatly enhanced by the conversion capabilities of the LFR fuel cycle. Because they incorporate a liquid coolant with a very high margin to boiling and benign interaction with air or water, LFR concepts offer substantial potential in terms of safety, design simplification, proliferation resistance and the resulting economic performance. An important factor is the potential for benign end state to severe accidents.

The LFR has development needs in the areas of fuels, materials performance, and corrosion control. During the next 5 years progress is expected on materials, system design, and operating parameters. Significant test and demonstration activities are underway and planned during this time frame.

1. Introduction

This paper reviews the past history of development of the LFR, summarizes the current status, and presents ongoing plans for future development. Past experience includes design development activities in several regions of the world as well as the significant deployment of a LFR technology in the Soviet Union for military (submarine propulsion) purposes. At present, the technical work underway by GIF participants includes activities associated with three different variants of the LFR representing three different systems sizes: the European Lead-cooled Fast Reactor (ELFR, 600 MWe); the Russian BREST-OD-300 (300 MWe); and the Small Secure Transportable Autonomous Reactor (SSTAR, 20 MWe) system concept designed in the US. Future activities include a variety of ongoing and planned efforts to address remaining technical issues while proceeding towards demonstration of modern LFR concepts.

In this paper, we present an overview and historical backdrop of LFR development, the present status of GIF activities related to the LFR, a summary of three reference LFR systems, a discussion of the advantages and challenges facing LFR development, and a summary of some considerations related to the safety attributes of LFRs under severe accident conditions in light of the Fukushima event.

The paper provides an update to the status of LFR development and GIF activities as presented at the GIF Symposium in San Diego (Alemberti et al., 2013a) in November 2012.

2. The historical backdrop of LFR development

The idea of fast reactors cooled by heavy liquid metals is an unfamiliar one to many, yet there is a considerable past history

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related to such reactors. In fact, the first fast reactor to operate was Clementine, a fast spectrum reactor cooled by the heavy liquid metal mercury. Clementine operated from 1946 to 1952 with a maximum output of 25 kWt (Smith and Cissei, 1978). After that time, operating experience with heavy liquid metal cooled reactors shifted to the Soviet Union/Russia while more recent design and experimental work has been carried out relatively broadly throughout the world. In 2002 the GIF identified the LFR as one of the six promising nuclear energy technologies to be considered for future advanced systems (NERAC and GIF, 2002). A GIF Provisional System Steering Committee (PSSC) was formed in 2005 and subsequently formalized as a System Steering Committee (SSC) through the initiation of a Memorandum of Understanding (MOU) in 2010.

2.1. Russian experience with LBE- and lead-cooled reactors

Significant industrial and operational experience with reactors cooled by lead or lead–bismuth eutectic (LBE) was gained by Russia (the Soviet Union and then the Russian Federation) in their programme to design, produce and deploy LBE-cooled reactors for submarine propulsion during the period from the mid 1960s until the 1990s. During this period, a total of 12 reactors and 15 reactor cores were built and deployed, including two reactors (and three of the reactor cores) that were operated onshore. In total, this programme represented about 80 reactor-years of operating experience (Walter et al., 2012).

Following the dissolution of the Soviet Union, interest in LFRs in the Russian Federation has remained strong. This interest is exemplified by some limited work devoted to accelerator driven subcritical (ADS) reactors cooled by LBE and, more importantly, two critical reactor concepts: the LBE-cooled SVBR (Svintsovo Vismutovyi Bystriy Reaktor) and the lead-cooled BREST (Bystriy Reaktor Estestrennoy Bezopasnosti). Both the SVBR and BREST developmental efforts are active with near-term (completion dates of 2018 and 2020) construction plans underway. In 2011, Russia joined the GIF SSC and signed the GIF-LFR-MOU.

2.2. European experience with LBE- and lead-cooled reactors

In Western Europe, initial efforts related to LFR development concentrated on Accelerator Driven Subcritical (ADS) systems for the transmutation of plutonium and minor actinides (MA). Key initiatives included EFIT (European Facility for Industrial Transmutation) and MYRRHA (Multi-purpose hybrid research reactor for high-tech applications) (De Bruyn et al., 2007). EFIT served as the starting point for the design of later critical reactor systems cooled by lead discussed below while MYRRHA continues as a major project intended to demonstrate both subcritical and critical operation of a system cooled by LBE while operating as a multipurpose irradiation facility.

In the process of developing these subcritical and critical lead-cooled systems, considerable effort has been spent to exploit to the greatest degree possible the inherent beneficial characteristics of lead or LBE as a coolant while introducing safety design as a primary consideration from the beginning.

The European research programme has been developed and funded primarily by the so-called Framework Programmes (FWP) of the European Community (EC). Starting with the 5th FWP in 1997, a consortium of major European organizations jointly initiated the development of an ADS prototype as part of the XT-ADS project funded by EC. The subject of transmutation was further investigated in the 6^{th} FWP starting in 2002 through the participation of several major industrial partners and research organizations in a project called IP-EUROTRANS. A major step for the LFR development was taken by the European Lead-cooled System (ELSY) project initiated in 2006. This project aimed to complete a conceptual design of a 600 MWe industrial size plant with challenging objectives in terms of compactness, economy and safety (Cinotti et al., 2008).

In 2010, the previous efforts and the experience gained in these projects were used to initiate two new projects as part of the 7th FWP: the CDT-FASTEF (Central Design Team-fast spectrum transmutation experimental facility) project and the LEADER (Lead-cooled European Advanced Demonstration Reactor) project.

The CDT-FASTEF project has spent the last three years conducting conceptual design of the MYRRHA facility, an ADS LBE-cooled facility that can be operated in both a subcritical as well as a critical mode, and is envisioned as a pilot plant for LFR technology.

The LEADER project concentrated its activities in the development of an enhanced concept for an industrial-sized critical reactor, the new configuration being designated ELFR (European Lead-cooled Fast Reactor) and, in parallel, undertook design activities for a smaller LFR demonstrator, sized at 120 MWe, designated as ALFRED, the Advanced Lead-cooled Fast Reactor European Demonstrator. Presently, strong efforts towards development of ALFRED are underway (Alemberti et al., 2012, 2013b).

A multiplicity of ancillary R&D projects was also initiated under FP7 to provide support to CDT-FASTEF and LEADER and to address a range of related issues of interest (and synergies) identified in the development of LBE and lead technologies. For example, some of these additional projects are: SARGEN-IV (Gen IV safety approach harmonization), SILER (Seismic-Initiated events risk mitigation in LEad-cooled Reactors), MATTER (Materials Testing and Rules), GETMAT (Gen IV and transmutation materials), THINS (Thermal-hydraulics of Innovative Nuclear Systems) and FREYA (Fast Reactor Experiments for hYbrid Applications).

In parallel with these R&D programmes, the LFR programme in Europe also benefits from 34 experimental facilities, in operation or under construction, in 10 European research institutions. While such facilities are not enumerated in detail in this paper, it is worthwhile to note that they are dedicated to the main issues identified by LFR designers and are distributed throughout Europe, testimonial to the great interest in numerous countries.

As far as GIF is concerned, Europe proposed and promoted the establishment of the GIF-LFR-MOU related to cooperation on LFR technology development.

2.3. Asian experience with LBE- and lead-cooled reactors

Japan and Korea have also conducted significant research into LFR reactors since at least the 1990s, while China only recently started its activities.

In 2010, Japan signed the GIF-LFR-MOU. Korea and China are presently participating in the LFR-PSSC activities as observers.

In Japan, several interesting projects have been pursued by the Tokyo Institute of Technology (Takahashi, 2012). The LSPR (LBE-cooled long-life safe simple small portable proliferation-resistant reactor) system is a small reactor with long-life core, a concept proposed in the early 1990s. This small reactor would be factory fabricated at an energy park, transported to its operating site, and operated for the reactor's life. The reactor would have a sealed vessel which would not be opened at the operating site for refuelling for reasons of proliferation resistance. At the end of the reactor life, it would be removed and replaced by a new one. The old reactor (with its expended fuel) would be shipped back to the nuclear energy park. There would be no residual radioactive waste left at the site. Thus, the operating site and host government or organization would not have to deal with spent fuel or radioactive waste from reactor fuelling operations.

In a separate effort, in 2004 the Tokyo Institute of Technology proposed the PBWFR (Pb–Bi-cooled direct contact Boiling Water Fast Reactor) design concept (Takahashi et al., 2005). This effort evaluated the feasibility of eliminating steam generators and primary pumps by direct injection of feedwater into hot LBE above the core to stimulate coolant circulation. The injected feedwater would boil in the reactor chimney, and steam bubbles would rise with buoyancy force. The resulting bubble motion would serve as the driving force of coolant circulation.

Another interesting concept, the CANDLE (Constant Axial shape of neutron flux, nuclide number densities and power shape) reactor, was also elaborated by the Tokyo Institute of Technology as a sodium-cooled design with a lead-cooled variant (Sekimoto and Nagata, 2008).

A significant Japanese effort relates to the development of LFR concepts by the Japan Nuclear Cycle Development Institute (JNC, presently JAEA) in their Feasibility Study on the Commercialized Fast Reactor Cycle (Takahashi, 2012). In Phase I of this study, typical fast reactor system concepts were identified and compared to different options: coolant types, including lead and lead–bismuth; plant size (i.e., large, medium, and small reactors); tank versus loop designs; and forced versus natural circulation cooling. In Phase II of the study, a concept of a lead–bismuth-cooled, medium size tank type, fast reactor with forced circulation was selected as the preferred LFR, and this concept was investigated to identify its attractive properties as well as drawbacks.

Finally, a notable effort was committed by the Central Research Institute for the Electric Power Industry (CRIEPI) and the Toshiba Corporation to develop the 4S reactor, an innovative small, long-life sodium-cooled reactor known as the 4S (Super Safe, Small and Simple) reactor. As part of this effort, some design work was also devoted to the consideration of a lead-cooled variant of this design, sometimes referred to as the L-4S.

Past LFR-related work in Korea has included efforts at the Seoul National University to develop concepts such as PEACER (Proliferation-resistant, Environment-friendly, Accident-tolerant, Continuable, and Economical Reactor) and BORIS (Kim et al., 2006; Hwang, 2006). Korea is still active from an experimental point of view with efforts focused mainly on thermal-hydraulics and corrosion experiments.

In 2009, the Chinese Academy of Sciences (CAS) started a new effort to develop an Accelerator Driven System (ADS) based on lead and LBE coolants. In 2011, CAS launched the Strategic Priority Research Program of “the Future Advanced Nuclear Fission Energy-ADS transmutation system”. The China LEAd-based Reactor (CLEAR) was selected as the reference reactor. The CLEAR reactor development plan includes three phases: a 10 MWth LBE-cooled research reactor (CLEAR-I) to be built in the 2010s, a 100MWth lead-based experimental reactor (CLEAR-II) in the 2020s and a 1000 MWth lead demonstration reactor (CLEAR-III) in the 2030s. In order to reach these goals, heavy liquid metal experimental facilities have been built with the aim of investigating the critical issues and key technologies of lead-based reactors, such as material issues, thermo-hydraulics, etc. The simulation tools for the LFR/ADS system are also being developed (Wu, 2013).

2.4. US experience with LBE- and lead-cooled reactors

Work on LFR concepts and technology in the U.S. has been carried out from 1997 to the present. During this time frame, work was carried out on lead corrosion and thermal-hydraulic testing at a number of different organizations and laboratories including Los Alamos National Laboratory (LANL), Argonne National Laboratory (ANL) and at the University of Nevada at Las Vegas (UNLV).

At the University of California at Berkeley (UC-B), design work was carried out on the Encapsulated Nuclear Heat Source (ENHS) and related design studies.

Of particular relevance is the development of the design of the small, secure transportable autonomous reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organizations over an extended period of time. This concept represents one of the three reference designs of the GIF-LFR Provisional System Steering Committee (PSSC) and, as such, is summarized further in a subsequent section of this paper (Smith et al., 2008).

It is also worth mentioning that some additional efforts have been carried out or are ongoing in the US. First, alloy and material development studies related to corrosion mitigation and, in particular, the development of the technology of Functionally Graded Composite materials – manufacturable materials that provide protection against corrosion in a molten lead or LBE environment – are being carried out at MIT (Short and Ballinger, 2010). In the industrial sector, companies such as Gen IV Energy and lake Chime PPRS are also pursuing LFR concepts for commercial application.

3. Current status and activities of the PSSC

The GIF-LFR Provisional System Steering Committee (PSSC) was initially formed in 2005. The original membership included the EC, the US, Japan and Korea. With Korea primarily in observer status between 2005 and 2008, this initial committee worked together to prepare a series of drafts of an initial LFR System Research Plan (LFR-SRP) (GIF-LFR PSSC, 2013), among its other activities.

During this first phase of the GIF research planning effort, beginning in 2005 and culminating in the completion of the final draft system research plan (SRP), two main directions or research thrusts were envisioned: the first was a (relatively) large central station plant for which the reference concept was ELSY (Cinotti et al., 2008); the second was a small transportable LFR system for which the reference concept was SSTAR (Smith et al., 2008).

In 2010, the GIF-LFR-MOU was signed between EC and Japan, and this resulted in a reformulation of the PSSC. Then in 2011, the Russian Federation added its signature to the MOU. In April 2012, the reformulated PSSC met in Pisa, Italy and a number of actions were defined. The US was invited to participate in the activities of PSSC as an observer, and the process of preparing a revised SRP was initiated. The new PSSC, with representatives of EC, Japan and Russia, envisioned various updates to the central station and small reactor thrusts while adding a mid-size LFR (i.e., the BREST-300) as a new thrust in the SRP. In addition, the PSSC decided to prepare a position paper describing the basic advantages and remaining research challenges of the LFR, to be posted on GIF website.

The second meeting of the reformulated PSSC took place on November 7–9, 2012 in Tokyo, hosted by the Tokyo Institute of Technology. This meeting was characterized by a high density of discussions between the members, especially on issues related to material corrosion and on the plant characteristics of the BREST-OD-300, which includes a site-dedicated fuel reprocessing plant.

The third meeting of the PSSC was hosted by OECD-NEA in Paris on March 7–8, 2013, in conjunction with the IAEA conference on Fast Reactors, FR-13. This most recent committee meeting saw an enlarged number of participants with additional observer representatives from China and Korea.

Both China and Korea have been accepted as observers of the GIF-PSSC activities and have declared the intention to proceed to LFR-MOU signature after discussion with respective government representatives.

The next meeting of the LFR-PSSC is scheduled for October 10–11, 2013 in Paris, hosted by OECD-NEA.

The reference concepts that form the basis of the revised SRP activities are the following.

- The European Lead Fast Reactor (ELFR) for the large, central station plant (600 MWe).
- The BREST-OD-300 (300 MWe) for the medium size plant.
- The small secure transportable autonomous reactor (SSTAR – 20 MWe) for the small system.

An overview description of each of these reference systems is provided in the next section.

4. Overview description of the three reference LFR systems

Beside obvious differences related to the size, the three systems taken as references for the GIF-LFR-PSSC activities share a significant number of technical issues and many common features, especially as far as safety design is concerned; thus, there are many commonalities from the point of view of design and engineering of these systems and the solutions adopted.

The ELFR system is an evolutionary design representing a modification to the earlier ELSY reactor concept. Fig. 1 provides an overview sketch of the ELFR reactor vessel and it contents.

Several of the relevant characteristics of the ELFR design are summarized below in Table 1.

The ELFR primary system has a pool-type configuration, with the main and safety vessels supported by a Y-support holding the main vessel in the upper part. The reactor vessel (RV) has been kept as compact as possible, in order to reduce the total coolant inventory and the corresponding seismic loads, while being of sufficient size to accommodate the required number of components (i.e., 8 steam generators (SGs), 8 primary pumps (PPs), and 8 decay heat dip coolers (DCs)).

The hot pool of the ELFR vessel is enclosed by an inner vessel (IV), connected to the PPs through suction pipes. Each PP is installed at centre of its corresponding SG, which transfers the heat from primary lead coolant to the water-steam in a superheated cycle. The free level of the hot pools inside each SG/PP unit is higher than the free level inside the inner vessel, the different heads depending on the pressure losses across component parts of the primary circuit.

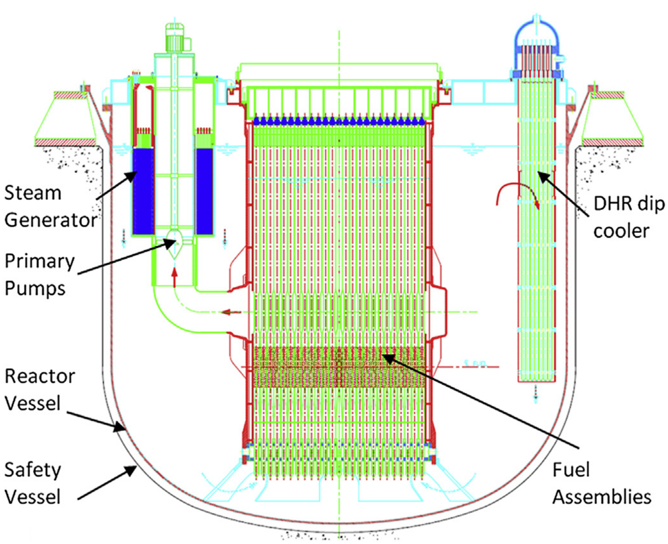

Fig. 1. ELFR, the European lead-cooled fast reactor.

Table 1
ELFR summary parameters.

Power (1500 MW (th))	600 WW(e)
Core diameter	4.5 m
Core height	1.4 m
Core fuel	MOX (1st load)
Coolant temp.	400/480 °C
Maximum clad temp.	550 °C
Net efficiency	~42%
Core breeding ratio –CBR	~1

The design is based on a core pressure loss of 0.9 bar and a total primary pressure loss of 1.4 bar.

The core inlet and outlet temperatures are 400 °C and 480 °C, respectively, allowing for a sufficient margin in the cold plenum from the freezing point of the lead coolant, while reducing the potential for embrittlement (for structures wetted by cold lead) and corrosion (for structures in hot molten lead).

The maximum speed of the primary coolant is specified at 2 m/s (10 m/s at the tips of the pump impeller blades) in order to limit erosion.

The internal reactor component arrangement and design presents a simple flow path for the primary coolant. The locations of the heat source (within the core) and of the heat sinks (SGs) allow for efficient natural circulation of the coolant under emergency shutdown conditions.

Two sets of safety systems for decay heat removal, one based on isolation condensers connected to the secondary system and the other based on dip coolers in the cold pool, have been considered as an integral part of the design from the beginning of the activities. Both systems are characterized by passive operation and redundancy while, in addition, being completely independent and diversified from one another.

The design of the core has been driven by the implementation of the so-called "adiabatic" (Artioli et al., 2010) reactor concept. The adiabatic reactor concept concerns the operation of a reactor with an equilibrium fuel cycle, so that the fuel composition remains the same between two successive loadings, ensuring the full recycling of all the actinides, with either natural or depleted uranium as only top-up/input material and fission products as well as reprocessing and fabrication losses as outputs, as illustrated below in Fig. 2.

This approach is conceptually very similar to that used for BREST-OD-300, presented in the next section.

4.1. The BREST-OD-300 Russian

4.1.1. Lead-cooled reactor

The BREST-OD-300 reactor is a pilot demonstration reactor (300 MWe) considered as a prototype of future commercial reactors of the BREST family for large-scale nuclear plants characterized by the idea of "natural safety." Fig. 3 provides an overview sketch of the BREST-OD-300 system.

Several of the relevant characteristics of the BREST design are summarized below in Table 2.

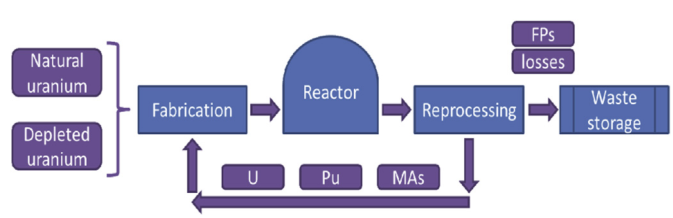

Fig. 2. ELFR adiabatic fuel cycle.

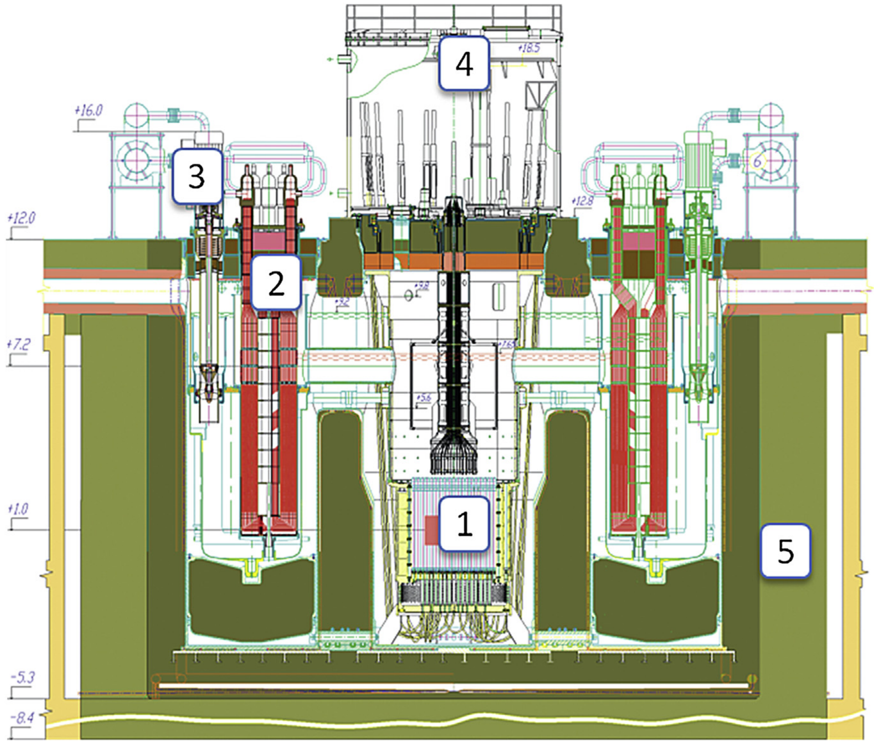

Fig. 3. BREST-OD-300.

BREST-OD-300 is a reactor facility of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation circuit with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault.

BREST has a widely spaced fuel lattice with a large coolant flow area, resulting in low pressure losses, favouring the establishment of primary natural circulation for decay heat removal. It shares with other designs the absence of uranium blankets, replaced by lead reflector with the proper albedo improving power distribution, providing a negative void and density coefficients, and ruling out the production of weapons-grade plutonium. The BREST decay heat removal systems are characterised by passive and time-unlimited residual heat removal directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere.

The fuel type considered for the first core of the BREST fast reactor is nitride of depleted uranium mixed with plutonium and Minor Actinides (MA), whose composition corresponds to that of irradiated (spent) fuel from PWR's following reprocessing and subsequent cooling for ~20 years.

The characteristics of lead allow for the operation with such fuel at an equilibrium composition. This mode of operation is characterized by full reproduction of fissile nuclides in the core (Core Breeding Ratio (CBR) ~1) with irradiated fuel reprocessing in the closed fuel cycle. Reprocessing is limited to the removal of fission products without separating Pu and minor actinides (MA) from the mix (U-Pu-MA). One of the notable characteristics of the BREST plant is that a reprocessing plant is co-located with the reactor, eliminating in principle any accident or problem due to fuel transportation.

Table 2
BREST summary parameters

Power 700 MW(th)	300 MW(e)
Core diameter	2.6 m
Core height	1.1 m
Core fuel	UN + PuN
Coolant temperature	420/540 °C
Maximum clad temperature	650 °C
Efficiency	43–44%
Core breeding ratio (CBR)	~1

4.2. The SSTAR, US small secure transportable autonomous reactor

SSTAR is a small modular reactor (SMR) that can supply 20 MWe/45 MWt with a reactor system that can be transported in a shipping cask. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life without refueling; and an innovative supercritical CO_2 (S-CO_2) Brayton cycle power conversion system. Fig. 4 provides an overview sketch of the SSTAR system.

Several of the relevant characteristics of the SSTAR design are summarized below in Table 3.

The present pre-conceptual design of SSTAR is that of a small shippable reactor (12 m × 3.2 m vessel), with a 15–30-year life open-lattice cassette core and large-diameter (2.5 cm) fuel pins held by spacer grids welded to control rod guide tubes.

The main mission of the 20 MWe (45 MWt) SSTAR is to provide incremental energy generation to match the needs of developing nations and remote communities without electrical grid connections, such as those that exist in Alaska or Hawaii, island nations of the Pacific Basin, and elsewhere.

Design features of the reference SSTAR in addition to the lead coolant, 15–30-year cassette core and natural circulation cooling, include autonomous load following without control rod motion, and use of a supercritical CO_2 (S-CO_2) Brayton cycle energy conversion system. The incorporation of inherent thermo-structural

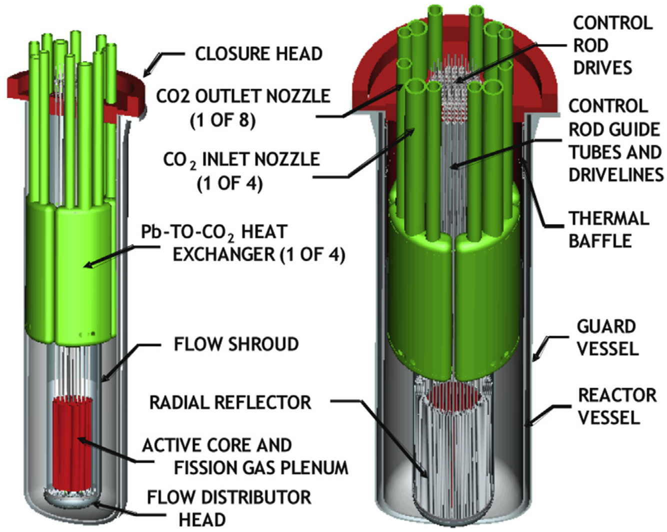

Fig. 4. SSTAR.

feedbacks imparts a high degree of passive safety, while the long-life cartridge core life imparts strong proliferation resistance.

5. Research challenges remain

Although many physical characteristics of lead used as a coolant constitute a set of clear advantages with respect to other potential reactor coolants, there are obviously some aspects that need specific developments. LFR research challenges are mainly related to the following aspects: the high melting point of lead; its opacity; coolant mass; and potential for corrosion of structural steels.

The high melting temperature of lead (327 °C) requires that the primary coolant system be maintained at sufficiently high temperatures to prevent solidification. This presents design as well as engineering challenges during operation and maintenance, although is not considered by designers a safety issue.

The opacity of lead, in combination with its high melting temperature, presents challenges related to inspection and monitoring of reactor in-core components as well as fuel handling. Important synergies are, however, possible with SFR technology, where specific developments are underway. In addition, design innovation can reduce needs in this area; for example, with the ELFR system, fuel element extension into the cover gas above the free surface of the coolant offers the possibility to directly monitor fuel status under more favourable conditions.

The high density and corresponding high mass of lead require careful consideration of structural and seismic design. This issue can be addressed by the adoption of technology such as seismic isolation as is being done in design-specific projects related to ELFR.

Significant challenges result from the phenomena related to lead corrosion of structural steels at high temperatures and flow rates. These phenomena require careful material selection and

Table 3
SSTAR summary parameters.

Power 45 MW(th)	20 MW(e)
Core lifetime	15–30 years
Core fuel	Nitride – N_{15} enriched
Coolant temperature	420/567 °C
Maximum clad temperature	650 °C
Efficiency	44%
Core breeding ratio (CBR)	~1

component and system monitoring during plant operations. In the past, Russian scientists developed the technology of continuous passivation of the structural steels based on their early LFR experience and were able to solve the problems encountered during early reactor deployments. Many efforts of a fundamental nature are being carried out in several European laboratories in order to investigate specific aspects and peculiarities of this technology.

6. LFR and extreme natural events

In the development of any reactor concept, safety is a critical consideration. Following the events related to the natural disaster and reactor accidents at Fukushima-Daiichi, it is especially important to consider the ability of reactor systems to respond successfully to extreme events.

The use of lead (or LBE) as a coolant offers several important advantages in this regard. In an interesting recent analysis, Toshinsky et al. (2011) considered the question of stored energy in various types of reactor coolants. This analysis highlights several important advantages of lead coolants with respect to the availability of energy (thermal, pressure-related or chemical) to exacerbate accident conditions whether initiated by natural phenomena or not. As low-pressure coolants with relative chemical inertness, lead and LBE have innate characteristics that enable a high level of passive safety and suggest more benign end states in the case of unforeseen accident conditions. In the following paragraphs, some particular aspects of LFR safety approaches and characteristics are identified.

Seismic and structural designs are important considerations for any reactor system, and especially larger systems. In the case of the current LFR system design for the largest reference system, ELFR, we note that the ability to respond to earthquakes has been significantly enhanced by the adoption of seismic isolation.

With respect to decay heat removal (DHR) systems, the current reference designs feature independent, redundant, diverse, and completely passive DHR systems. Only actuation (through valve alignment) is active, and this would use local stored energy. As a result, station blackout – which was a critical factor at the Fukushima-Daiichi accidents – would not present a threat; any initiating event would be managed without requiring AC power or other external energy source.

Even if one postulates station blackout without functional DHR systems available, recent safety analyses for ELFR demonstrated that fuel and cladding temperatures would not reach critical levels. In fact, safety analyses are now normally performed assuming the absence of off-site power.

A complete core melt would be extremely unlikely due to favorable intrinsic lead characteristics: its high thermal inertia and very high boiling point would limit the possibility for fuel melting, and the higher density of the coolant in contrast with the oxide fuel would result in fuel dispersion in contrast to fuel compaction if an otherwise unforeseen event resulted in fuel disruption.

Furthermore, in the very unlikely event of an extreme Fukushima-like scenario (or beyond) leading to the loss of all heat sinks (i.e., loss of both DHR and secondary systems), reactor heat could still be removed by water cooling of the cavity between the reactor primary and safety vessels, while in the extreme case of a reactor primary vessel failure, additional systems external to the safety vessel can be envisioned to cool the lead pool. This scenario would imply a highly extreme situation in which all heat sinks of the system had been already lost. A main advantage of the LFR is that in such an extreme, beyond design condition a highly abundant and readily available fluid (i.e., water) can be used to cool the reactor and retain the system in a safe condition. The benign behaviour of the LFR in response to such extreme (and unanticipated) events is enabled by the inherent, natural characteristics of the coolant and is not dependent on complex or engineered features that could be subject to failure.

Thus, the reactor systems envisioned as references for the GIF-LFR-PSSC research activities present highly promising behaviour when challenged by extreme events. The physical and chemical characteristics of lead as a coolant provide very advantageous safety feedbacks that have been leveraged by designers to further enhance system response to any envisaged transient.

7. Conclusions

In conclusion, lead- and LBE-cooled systems offer great promise in terms of fast reactor plant simplification, performance and safety response while offering sustainability advantages common to other fast reactor systems.

The Russian experience with the deployment of LBE-cooled systems for submarine propulsion provided an excellent demonstration that the LFR can be produced and operated on an industrial scale.

Still, additional work is needed to achieve commercial deployment of new commercial LFR systems. Some important areas for R&D include the following.

- Completion of designs of commercializable systems as well as demonstration systems.
- Testing of special materials for use in lead environment.
- Completion of fuel studies, including recycle.
- Special studies (e.g., studies related to seismic response; sloshing; LBE dust/slag formation).
- Evaluation of long term radioactive residues from fuel and system activation.
- Technology pilot plant/demo activities.

Finally, in the post-Fukushima environment, the unique safety potential of the LFR should be recognized and leveraged.

Nomenclature

ADS	accelerator driven subcritical
ALFRED	advanced lead-cooled fast reactor European demonstrator
ANL	Argonne National Laboratory
BREST	Bystriy Reaktor Estestrennoy Bezopasnosti
CANDLE	constant axial neutron flux, densities and power shape during life of energy
CBR	core breeding ratio
CDT-FASTEF	Central Design Team for a fast spectrum transmutation experimental facility
CRIEPI	Central Research Institute for the Electric Power Industry
DC	decay heat dip cooler
DHR	decay heat removal
EC	European community
EFIT	European facility for industrial transmutation
ELSY	European lead-cooled system
ELFR	European lead-cooled fast reactor
FA	fuel assembly
FREYA	fast reactor experiments for hybrid applications
FWP	framework programme
GETMAT	Gen IV and transmutation materials
GIF	Generation IV International Forum
IV	inner vessel
JAERI	Japan Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LBE	lead–bismuth eutectic
LEADER	lead-cooled European advanced demonstration reactor
LFR	lead-cooled fast reactor
LLNL	Lawrence Livermore National Laboratory
LSPR	LBE-cooled long-life safe simple small portable proliferation-resistant reactor
ISI	in-service inspection
MA	minor actinide
MATTER	materials testing and rules
MIT	Massachusetts Institute of Technology
MOU	memorandum of understanding
MYRRHA	multi-purpose hybrid research reactor for high-tech applications
NERAC	U.S. Nuclear Energy Research Advisory Committee
PBWFR	Pb–Bi-cooled direct contact boiling water fast reactor
PEACER	proliferation-resistant, environment-friendly, accident-tolerant, continuable, and economical reactor
PP	primary pump
PSSC	Provisional System Steering Committee
RV	reactor vessel
RVACS	reactor vessel air cooling system
SARGEN-IV	Gen IV safety approach harmonization
SG	steam generator
SGTR	steam generator tube rupture
SILER	seismic-initiated events risk mitigation in lead-cooled reactors
SMR	small modular reactor
SRP	system research plan
SSC	System Steering Committee
SSTAR	small secure transportable autonomous reactor
SVBR	Svintsovo Vismutovyi Bystriy Reaktor
THINS	thermal-hydraulics of innovative nuclear systems
UNLV	University of Nevada at Las Vegas

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